UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
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NOV 19 2009

Randall K. Edington, Executive
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SUBJECT: PALO VERDE NUCLEAR GENERATING STATION – NRC COMPONENT DESIGN BASES INSPECTION REPORT 05000528; 05000529; 05000530/2009008

Dear Mr. Edington:

On July 16, 2009, the U.S. Nuclear Regulatory Commission (NRC) completed a component design bases inspection at your Palo Verde Nuclear Generating Station. The enclosed report documents our inspection findings. The preliminary findings were discussed on July 16, 2009, with Mr. Hesser and members of your staff. After additional in-office inspection, a final telephonic exit meeting was conducted on October 8, 2009, with Mr. Bement and others of your staff.

The inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. The team reviewed selected procedures and records, observed activities, and interviewed cognizant plant personnel.

Based on the results of this inspection, the NRC has identified three findings that were evaluated under the risk significance determination process. Violations were associated with three of the findings. All three of the findings were found to have very low safety significance (Green) and the violations associated with these findings are being treated as noncited violations, consistent with Section VI.A.1 of the Enforcement Policy. If you contest any of the noncited violations, or the significance of the violations you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with copies to the Regional Administrator, U.S. Nuclear Regulatory Commission, Region IV, 612 East Lamar Blvd., Suite 400, Arlington, Texas 76011; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the Palo Verde Nuclear Generating Station. In addition, if you disagree with the characterization of any finding in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, Region IV, and the NRC Resident Inspector at Palo Verde Nuclear Generating Station. The information you provide will be considered in accordance with Inspection Manual Chapter 0305.

In accordance with Code of Federal Regulations, Title 10, Part 2.390 of the NRC's Rules of Practice, a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

Ray Kellar

Ray Kellar, P.E., Chief
Engineering Branch 1
Division of Reactor Safety

Docket: 50-528; 50-529; 50-530
License: NPF-41; NPF-51; NPF-74

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Component Design Basis Inspection
Document Request

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U.S. NUCLEAR REGULATORY COMMISSION

REGION IV

Docket:	50-528; 50-529; 50-530
License:	NPF-41; NPF-51; NPF-74
Report Nos.:	05000528, -529, -530/2009008
Licensee:	Arizona Public Service Company
Facility:	Palo Verde Nuclear Generating Station
Location:	5951 S. Wintersburg Road Tonopah, Arizona
Dates:	June 22-26, 2009 July 6-16, 2009
Team Leader:	S. Makor, Senior Reactor Inspector, Engineering Branch 1
Inspectors:	K. Clayton, Senior Reactor Inspector, Engineering Branch 1 M. Young, Reactor Inspector, Engineering Branch 1 S. Alferink, Reactor Inspector, Engineering Branch 2
Accompanying Personnel:	G. Skinner, Electrical Contractor C. Baron, Mechanical Contractor
Others:	M. Runyan, Senior Reactor Analyst
Approved By:	Ray Keller, P.E., Chief Engineering Branch 1 Division of Reactor Safety

SUMMARY OF FINDINGS

IR 05000528; -529; -530/2009008; June 22-26, 2009 and July 6-16, 2009; Palo Verde Nuclear Generating Station: baseline inspection, NRC Inspection Procedure 71111.21, "Component Design Bases Inspection."

The report covers an announced inspection by a team of four regional inspectors and two contractors. Three findings and one unresolved item were identified. All of the findings were of very low safety significance. The final significance of most findings is indicated by their color (Green, White, Yellow, or Red) using Inspection Manual Chapter 0609, "Significance Determination Process." Findings for which the significance determination process does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 4, dated December 2006.

A. NRC-Identified Findings

Cornerstone: Mitigating Systems

- Green. The team identified a Green noncited violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," for the failure to perform an adequate operability evaluation for the condensate storage tank as required by site procedures. Specifically, upon discovery of the condition, the licensee performed an immediate operability determination evaluation based on concerns with the capability of the loop seal to provide protection from vacuum conditions. Subsequently, the licensee performed additional assessments of their overall program which included the specified operability evaluation in a component design bases review and closure of a confirmatory action letter and failed to identify the inadequacy. During the inspection, the team reviewed the operability determination and identified that the licensee failed to consider or identify concerns with the ability of the condensate storage tank pressure relief valves to operate after a design basis earthquake. The licensee entered this issue into their corrective action program as Palo Verde Action Request 3353683.

  This finding is more than minor because it is associated with the protection against external events (seismic) attribute of the Mitigating Systems Cornerstone and adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The risk significance of this finding was determined using Inspection Manual Chapter 0609, Attachment 4, "Phase 1 – Initial Screening and Characterization of Findings." The finding is of very low safety significance (Green) since the finding did not result in a loss of operability, a loss of system safety function, an actual loss of safety function of a single train for greater than its technical specification allowed outage time, or an actual loss of safety function for greater than 24 hours and the finding did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event. This finding has a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program since the licensee failed to properly evaluate for operability [P.1(c)] (Section 2.16).

- Green. The team identified a Green noncited violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," with programmatic implications for the licensee's failure to follow site procedures and incorporate updated vendor information for the reactor trip breakers. Specifically, the licensee failed to incorporate an updated revision of the maintenance program manual and at least two technical bulletins from the reactor trip breaker vendor. The licensee entered this issue into their corrective action program as Palo Verde Action Requests 3354252 and 3355082.

  This finding is more than minor because it is associated with the equipment performance attribute of the Mitigating Systems Cornerstone and adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The risk significance of this finding was determined using Inspection Manual Chapter 0609, Attachment 4, "Phase 1 – Initial Screening and Characterization of Findings." The finding is of very low safety significance (Green) since the finding did not result in a loss of operability, a loss of system safety function, an actual loss of safety function of a single train for greater than its technical specification allowed outage time, or an actual loss of safety function for greater than 24 hours and the finding did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event. This finding has a crosscutting aspect in the area of problem identification and resolution associated with operating experience since the licensee failed to implement changes to station processes, procedures, equipment, and training programs [P.2(a)] (Section 3.2).

- Green. The team identified a noncited violation of very low safety significance for failure to effectively implement the corrective action requirements of Regulatory Guide 1.155, "Station Blackout," Appendix A, Criterion 8, "Corrective Action," which were adopted by the licensee in order to meet 10 CFR 50.63, "Loss of All Alternating Current." Although the licensee started a vault monitoring program for water intrusion in vaults with electrical cables in 2003, the effort to prevent exposure of medium voltage cables to submerged conditions has been ineffective for certain vaults that contain the 15kV station blackout generator output cables. Additionally, there are 27 splices in these cables which have contributed to cable test failures in previous meggar resistance tests that, in some cases, required splice replacement in order to pass resistance tests. The licensee entered this issue into their corrective action program as Palo Verde Action Requests 3350712, 3350713, 3350939, and 3352858.

  This finding is more than minor because it is associated with the design control and equipment performance attribute of the Mitigating Systems Cornerstone and adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The risk significance of this finding was determined using Inspection Manual Chapter 0609, Attachment 4, "Phase 1 – Initial Screening and Characterization of Findings." The finding is of very low safety significance (Green) since the finding did not result in a loss of operability, a loss of system safety function, an actual loss of safety function of a single train for greater than its technical specification allowed outage time, or an actual loss of safety function for greater than 24 hours and the finding did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event. The finding was reviewed for crosscutting aspects and none were identified (Section 3.5).

B. Licensee-Identified Violations.

None were identified.

REPORT DETAILS

1 REACTOR SAFETY

Inspection of component design bases verifies the initial design and subsequent modifications and provides monitoring of the capability of the selected components and operator actions to perform their design bases functions. As plants age, their design bases may be difficult to determine and important design features may be altered or disabled during modifications. The plant risk assessment model assumes the capability of safety systems and components to perform their intended safety function successfully. This inspectable area verifies aspects of the Initiating Events, Mitigating Systems and Barrier Integrity Cornerstones for which there are no indicators to measure performance.

1R21 Component Design Bases Inspection (71111.21)

The team selected risk-significant components and operator actions for review using information contained in the licensee's probabilistic risk assessment. In general, this included components and operator actions that had a risk achievement worth factor greater than 1.3 or a Birnbaum value greater than 1E-6.

a. Inspection Scope

To verify that the selected components would function as required, the team reviewed design basis assumptions, calculations, and procedures. In some instances, the team performed calculations to independently verify the licensee's conclusions. The team also verified that the condition of the components was consistent with the design bases and that the tested capabilities met the required criteria.

The team reviewed maintenance work records, corrective action documents, and industry operating experience records to verify that licensee personnel considered degraded conditions and their impact on the components. For the review of operator actions, the team observed operators during simulator scenarios, as well as during simulated actions in the plant.

The team performed a margin assessment and detailed review of the selected risk-significant components to verify that the design bases have been correctly implemented and maintained. This design margin assessment considered original design issues, margin reductions because of modifications, and margin reductions identified as a result of material condition issues. Equipment reliability issues were also considered in the selection of components for detailed review. These included items such as failed performance test results; significant corrective actions; repeated maintenance; 10 CFR 50.65(a)1 status; operable, but degraded, conditions; NRC resident inspector input of problem equipment; system health reports; industry operating experience; and licensee problem equipment lists. Consideration was also given to the uniqueness and complexity of the design, operating experience, and the available defense in-depth margins.

The inspection procedure requires a review of 20 to 30 total samples that include 10 to 20 risk-significant and low design margin components, 3 to 5 relatively high-risk operator actions, and 4 to 6 operating experience issues. The sample selection for this inspection was 16 components, 5 operator actions, and 5 operating experience items.

.2 Results of Detailed Reviews for Components:

.2.1 High Pressure Safety Injection Pump

a. Inspection Scope

The team reviewed the Updated Final Safety Analysis Report, system design criteria, current system health report, selected drawings, operating procedures, inservice testing results, past corrective action documents and condition reports, calculations, and operability determinations. The inspection team also performed walkdowns of the pump and performed interviews with design and system engineering personnel to ensure the capability of this component to perform its required post-accident function. This review included system hydraulic and net positive suction head analyses for post accident operation. The inspection team also reviewed the status of a refueling water tank vortexing issue, which had been originally identified in October 2005 during a previous NRC inspection.

b. Findings

No findings of significance were identified

.2.2 Low Pressure Safety Injection Pump

a. Inspection Scope

The team reviewed the Updated Final Safety Analysis Report, system design criteria, current system health report, selected drawings, operating procedures, inservice testing results, past corrective action documents and condition reports, calculations, and operability determinations. The inspection team also performed walkdowns of the pump and performed interviews with design and system engineering personnel to ensure the capability of this component to perform its required post-accident function. This review included system hydraulic and net positive suction head analyses for post accident operation.

b. Findings

No findings of significance were identified.

.2.3 Main Steam Atmospheric Dump Valves

The team reviewed the Updated Final Safety Analysis Report, system design criteria, current system health report, selected drawings, operating procedures, inservice testing results, past corrective action documents and condition reports, calculations, and operability determinations. Specifically, the inspection team reviewed the capability and capacity of the plant instrument air and nitrogen systems to operate these valves under various plant conditions, including loss of off-site power, station blackout, and accident conditions. The inspection team also reviewed the operating temperature and environment qualification of the valves. The inspection team performed walkdowns of the valves and performed interviews with design and system engineering personnel to ensure the capability of this component to perform its required function.

b. Findings

No findings of significance were identified.

.2.4 Containment Sump Isolation Valve (1JSIBUV0675)

a. Inspection Scope

The team reviewed the Updated Final Safety Analysis Report, system design criteria, current system health report, selected drawings, operating procedures, inservice testing results, past corrective action documents and condition reports, calculations, and operability determinations. Specifically, the team inspected the applicable motor-operated valve (MOV) calculations, testing procedures and results, and modifications related to the valve to verify its capability to operate under post-accident conditions. The team reviewed electrical calculations to verify the appropriate voltage values were included in the valve calculations. The team also reviewed operating procedures related to the valve to ensure they were consistent with the design basis calculations and the licensing basis.

b. Findings

No findings of significance were identified.

.2.5 Unit 1 Main Feedwater Pumps

a. Inspection Scope

The team reviewed the Updated Final Safety Analysis Report, system design criteria, current system health report, selected drawings, operating procedures, inservice testing results, past corrective action documents and condition reports, calculations, and operability determinations.

Specifically, the review was performed to verify the reliability of the feedwater pumps during normal and transient conditions. The inspection team performed walkdowns of the pumps and performed interviews with design and system engineering personnel to ensure the capability of the pumps to perform their required function.

b. Findings

No findings of significance were identified.

.2.6 Turbine Driven Auxiliary Feedwater Pump (2AFA-P01)

a. Inspection Scope

The team reviewed the Updated Final Safety Analysis Report, system design criteria, current system health report, selected drawings, operating procedures, inservice testing results, past corrective action documents and condition reports, calculations, and operability determinations. Specifically, the team reviewed an operability determination which credited the non-essential (nonsafety safety-related) motor-driven auxiliary feed water pump during shutdown cooling conditions. The team verified the reliability of the turbine driven auxiliary pump during normal and transient conditions. The team reviewed the net positive suction head for the pump and the change in bearing lubrication type. The team also reviewed vendor recommendations to ensure that the licensee was operating equipment appropriately.

b. Findings

No findings of significance were identified.

.2.7 Turbine Driven AFW Pump Discharge Check Valve (1PAFA-V015)

a. Inspection Scope

The team reviewed the Updated Final Safety Analysis Report, system design criteria, current system health report, selected drawings, inservice testing results and past corrective action documents. Specifically, the team reviewed inservice testing basis criteria and the open and closure tests of the valve to ensure that the operability of the essential motor-driven auxiliary feed water pump when the turbine-driven auxiliary feed water pump is secured. The team also reviewed vendor recommendations to ensure that the licensee was operating equipment appropriately.

b. Findings

No findings of significance were identified.

.2.8 Diesel Generator (3DGA), Intercooler (3MDGAE01A), Jacketwater Cooler (3MDGAE05)

a. <u>Inspection Scope</u>

The team reviewed the Updated Final Safety Analysis Report, system design criteria, current system health report, selected drawings, operating procedures, inservice testing results and past corrective action documents. Specifically, the team reviewed the design basis heat load sizing analysis for the heat exchangers to verify its capability to meet design basis heat removal requirements. The team reviewed an operability determination pertaining to cap screw defects that were found on the diesel generator fuel oil injector pumps. The vendor of the cap screws identified a subset of screws which contained the defects; therefore the licensee took prompt action and has since replaced all of the cap screw defects. A review of NRC Generic Letter 89-13 program requirements for thermal performance testing, chemistry control, maintenance and corrective actions was conducted. The team also reviewed vendor recommendations to ensure that the licensee was operating equipment appropriately.

b. <u>Findings</u>

No findings of significance were identified.

.2.9 <u>Diesel Fuel Oil Pump (3MDFA-P01)</u>

a. <u>Inspection Scope</u>

The team reviewed the Updated Final Safety Analysis Report, system design criteria, current system health report, selected drawings, operating procedures, inservice testing and past corrective action documents. Specifically, the team reviewed the effects of external missiles on the diesel fuel oil storage tank vent and the fill lines to ensure that the pump would not create a vacuum in the tank and render the pump inoperable. The team reviewed the net positive suction head for the pump to ensure operability of the pump, as well as the fuel oil chemistry procedures. The team also reviewed vendor recommendations to ensure that the licensee was operating equipment appropriately.

b. <u>Findings</u>

No findings of significance were identified.

.2.10 <u>Diesel Generator, Electrical (3DGA)</u>

a. <u>Inspection Scope</u>

The team reviewed the Updated Final Safety Analysis Report, system design criteria, current system health report, selected drawings, operating procedures, inservice testing and past corrective action documents. Specifically, the team reviewed corrective actions to determine whether they adequately addressed the causes and to determine whether they were being implemented in a timely manner. The team also reviewed significant condition report/disposition request (CRDR) 3285804, "2008 Significant CRDRS - High Incidence of Electrical and Electronic Safety Related System Failures" and interviewed members of the diesel generator High Intensity Team to determine whether ongoing diagnostic and corrective actions are appropriate and effective at preventing future failures. The team reviewed calculation 01-EC-MA-0221 to assess diesel generator loading margins and performed a walkdown of the diesel generator to assess material condition, and the presence of hazards. This also included an assessment of building ventilation and susceptibility of diesel generator support systems to damage from tornado depressurization.

b. Findings

No findings of significance were identified.

.2.11 Inverter (2E-PNA –N11)

a. Inspection Scope

The team reviewed the Updated Final Safety Analysis Report, system design criteria, current system health report, selected drawings, operating procedures, inservice testing and past corrective action documents. Specifically, the team's review focused on corrective actions related to significant CRDR 3165478, "Multiple 120VAC Vital Inverters Events." The team reviewed the history of inverter failures and anomalies, and the licensee's evaluations. This included a review of cause determinations to determine whether they adequately addressed the issues. The team reviewed corrective actions to determine whether the adequately addressed the causes and to determine whether they were being implemented in a timely manner. Since the cause of some anomalies has not yet been determined, the team reviewed ongoing monitoring and diagnostic measures to determine whether they adequately addressed the potential causes. The team reviewed the inverter vendor manuals, and industry guidance to determine whether preventive maintenance measures were adequate. The team performed a review of corrective action and maintenance histories to determine whether all the issues had been correctly captured and characterized, and to determine whether adverse trends were continuing. The team reviewed modification 3E-PN-013 for the installation of static transfer switches to the Units 2 and 3 inverters, to determine whether the evaluation adequately considered failure modes and effects during LOOP scenarios. The team performed a walkdown of the equipment to assess material condition and the presence of hazards.

b. Findings

No findings of significance were identified.

.2.12 Circuit Breakers

a. Inspection Scope

The team reviewed the Updated Final Safety Analysis Report, system design criteria, current system health report, selected drawings, operating procedures, inservice testing and past corrective action documents. Specifically, the team reviewed recovery plan "CB" to determine whether issues with circuit K-Line breaker overhaul schedules and GE TEC circuit breaker coordination were being adequately resolved. The team discussed overhaul schedules for the K-Line breakers with the responsible component engineer and reviewed corrective action records to determine whether the proposed schedules were adequate to provide reasonable assurance of operability. The team reviewed the progress of modification PH-265 to address GE TEC breaker coordination issues to determine whether the scheduled completion dates were adequate. The team also reviewed corrective action history for circuit breakers to determine whether there were any other design or maintenance issues besides the ones address in the recovery plan.

b. Findings

.2.13 Essential Spray Pond Pump House Ventilation Fan

a. Inspection Scope

The team reviewed the Updated Final Safety Analysis Report, system design criteria, current system health report, selected drawings, operating procedures, inservice testing and past corrective action documents. Specifically, the team performed a limited review of this component and focused on the evaluation of undervoltage concerns that were performed under engineering evaluation request (EER) 93-XE-001. The team reviewed the EER to determine whether it applied conservative inputs, assumptions, and methodologies, and whether the results of the analyses and testing justified the conclusions. The team reviewed corrective action and maintenance history for the motor to determine whether there had been any occurrences that would affect the original conclusions of the EER. In addition, the team assessed margins associated with the analyses and tests to determine whether they afforded reasonable assurance of operability under worst case conditions.

b. Findings:

No findings of significance were identified.

.2.14 4.16 kV Emergency Bus 2G

a. Inspection Scope

The team reviewed the Updated Final Safety Analysis Report, system design criteria, current system health report, selected drawings, operating procedures, inservice testing and past corrective action documents. Specifically, the team reviewed AC load flow calculations to determine whether the 4160V system had sufficient capacity to support its required loads under worst case accident loading and grid voltage conditions. The team reviewed the design of the 4160V bus degraded voltage protection scheme to determine whether it afforded adequate voltage to safety related devices at all voltage distribution levels. This included a review of degraded voltage relay setpoint calculations, motor starting and running voltage calculations, MOV calculations, and motor control center control circuit voltage drop calculations. The team reviewed corrective action documents and maintenance records to determine whether there were any adverse operating trends. The team reviewed operating procedures to determine whether the limits and protocols for maintaining offsite voltage were consistent with design calculations. In addition, the team performed a walkdown of the 4160V safety buses to assess operability and material condition.

b. Findings

Failure to Perform Adequate Calculations to support the Degraded Voltage Relay Setpoint

Introduction. The team identified an unresolved item (URI) with two aspects, relating to the degraded voltage protection scheme. The first aspect involved the time delay for the degraded voltage protection scheme. Specifically, the time delay of 35 seconds for transfer of safety buses to the onsite power supplies may be too long to prevent core damage in case of a sustained degraded voltage condition concurrent with an accident. The second aspect involves the lack of calculations to support the degraded voltage relay voltage setpoint.

Description. There has been a series of correspondence between the NRC and Palo Verde regarding degraded voltage relays and the team noted during the inspection that item 32 of NRC letter dated December 12, 1977 provided guidance on complying with NRC requirements for the capacity and capability of power sources. This letter required the installation of automatic voltage monitors to detect the presence of a sustained degraded voltage condition. Position 1.a of item 32 stated that the selection of voltage and time setpoints shall be determined from an analysis of the voltage requirements of the safety-related loads at all onsite system distribution levels, and position 1.C(1) stated, "The time delay selected shall not exceed the maximum time delay that is assumed in the FSAR accident analysis." The licensee's proposals for meeting these requirements were accepted by the NRC in NUREG-0857, Supplement 5, dated November 1983. In 1992, the NRC Electrical Distribution System Functional Inspection (EDSFI) team identified a lack of calculations to support the setpoints of degraded voltage relays. Palo Verde updated the voltage calculations, and identified several deficient conditions involving the degraded voltage scheme in LER 93-011 and its supplements. During the period 1993 to 1999, Palo Verde and the NRC exchanged several communications, including two license amendment requests, which discussed these issues. License amendment request dated December 16, 1998, proposed technical specifications changes involving administrative controls aimed at preventing the spurious separation of safety buses during an accident. Specifically, Technical

Specification 3.8.1, Condition G provided an 1-hour action statement for restoring the capability of an offsite power source that was determined not to be capable, or to transfer of the affected safety buses to the onsite emergency diesel generators. The technical specification bases were revised to provide guidance and criteria for determining the capability of an offsite source.

The first issue identified by the team involved the degraded voltage relay time delay. The time delay for the degraded voltage relay is set at ≥ 28.6 seconds and ≤ 35 seconds. This time delay could result in a delay in supplying water to the core in case of an accident concurrent with degraded voltage, due to the inability of electrical equipment to respond as required during the timeout period. Final Safety Analysis Report 6.3.3.3.2 states that the accident analysis assumes 30 seconds from a safety injection actuation signal (SIAS) to the time that safety injection flow is delivered to the core.

As part of a December 16, 1998 submittal, Palo Verde included a revision to Final Safety Analysis Report Section 8.3.1.1.3.13 to delete criteria associated with the degraded voltage relay time delay taken from the December 12, 1977, letter, item 32, position 1.C(1), that is quoted above. This was done without explanation or any apparent relationship to the primary objective of the amendment request. The NRC questioned this deletion in item 13 of a letter dated June 14, 1999. The Palo Verde response provided in a letter dated July 16, 1999, stated that the degraded voltage relays were not capable of providing a protective function, or required to function during a design basis accident, so that the time delay had no effect on the accident analysis. The safety evaluation report dated December 29, 1999, did not comment on this response and there was no indication that the NRC had accepted the assertion that a degraded voltage condition concurrent with an accident was not credible. The team also noted that the degraded voltage relay delay setpoints are still included in Final Safety Analysis Report Table 7.3-1B, "Engineered Safety Features Response Times," and Table 7.3-11A, "NSSS Engineered Safety Features Actuation System Setpoints and Margins to Actuation." They are also included in Technical Specification 3.3.7, "Diesel Generator (DG) — Loss of Voltage Start (LOVS)."

The team questioned the rationale included in the July 1999 licensee response, as follows:

- The basis for the original time delay of ≤ 35 seconds did not appear to be based on the correct criteria. It was based on the time required for starting a reactor coolant pump motor rather than the much shorter time required for starting a large emergency core cooling system motor.

- A shorter time delay for accident conditions can be implemented without increasing the risk of spurious separation for momentary voltage excursions, or for longer voltage excursion during normal operation.

- A shorter time delay will not delay the time required to provide water to the core, but will actually improve it.

- The licensing basis for degraded voltage concurrent with an accident was not properly characterized.

- The licensee did not provide any information indicating that a shorter time delay would increase the probability of double sequencing, or worsen its affects.

The second issue identified by the team is in regards to the calculations for the degraded voltage relay setpoints. In the Technical Specification 3.3.7, "Diesel Generator (DG)-Loss of Voltage Start (LOVS)" a degraded voltage relay trip setpoint of ≥ 3697 V and ≤3786 V is specified and the team noted that the current calculations of record do not establish the adequacy of this setpoint. The load flow calculation 01-EC-MA-0221, "AC Distribution", was used as an input to other calculations which analyzed voltage available to safety-related loads based on the minimum voltage afforded by administrative controls described in Technical Specification 3.8.1, Condition G which provided results that were higher than that afforded by the automatic protection of the degraded voltage relays by up to approximately 2.5 percent.

The lack of calculations to support the setpoints was previously documented by the licensee in Palo Verde Action Request (PVAR) 3253612 and CRDR 3256729. In the operability determination PVAR, it stated that this was a "paper nonconformance" that does not affect operability because it is not credible to have switchyard voltage low enough to cause degraded voltage due to technical specification limitations provided for in Technical Specifications 3.8.1 Condition G. The team performed a review of the record relating to the original licensing of the degraded voltage relays in the early 1980s and to the license amendment request dated December 16, 1998, relating to the Technical Specification 3.8.1 change that implemented administrative controls to limit the vulnerability to spurious grid separation. The licensee had offered the proposition that degraded voltage concurrent with an accident was not credible, but the team could not find evidence that the NRC had accepted this position, or that the degraded voltage relays were no longer required to perform a protective function during accidents. The team believes that the licensee's position in the operability determination was further refuted by the continued inclusion of the degraded voltage relay setpoint in FSAR Table 7.3-11A, and Technical Specification 3.3.7.

Although the licensee disputed the need to perform calculations based on the degraded voltage relay setpoint, preliminary calculations were performed by the licensee to assess the adequacy of voltage based on the settings of the degraded voltage relays. These calculations showed that motor control center circuits would generally have adequate margin to operate at the lowest voltages afforded by the degraded voltage relay setpoint, although available voltage margins for several devices were reduced by approximately 75 percent. Evaluation of MOV circuits was ongoing at the close of the inspection.

Because the licensee has taken the position that formal calculations to support the design basis for the degraded voltage relays need not be performed, this item is considered an URI pending further review by NRR and is identified as : URI 05000528;

05000529; 05000530/2009008-01, "Failure to Perform Adequate Calculations for Degraded Voltage Relay Setpoints."

.2.15 Auxiliary Feedwater Steam Admission Valve (SGA-UV-134A)

a. Inspection Scope

The team reviewed the Updated Final Safety Analysis Report, design and licensing bases, operating procedures, calculations, modifications, inservice test data, and corrective action documents to verify the capability of the auxiliary feedwater steam admission valve to perform its safety function. Specifically, the team reviewed the modification that replaced the solenoid operator with a motor operator as well as the in-service test procedures, data, and acceptance criteria associated with the motor operator.

b. Findings

No findings of significance were identified.

.2.16 Condensate Storage Tank (CTE-T01)

a. Inspection Scope

The team reviewed the Updated Final Safety Analysis Report, design and licensing bases, operating procedures, calculations, and corrective action documents to verify the capability of the condensate storage tank to perform its safety function. Specifically, the team reviewed the capability of the pressure relief valves to protect the condensate storage tank from overpressure and vacuum conditions during design basis events. The team also reviewed the bases for the tank capacity and alarm setpoints as well as the analysis of instrument uncertainty.

b. Findings

Failure to Identify an Inadequate Operability Evaluation for the Condensate Storage Tank

Introduction. The team identified a Green noncited violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," for the failure to perform an adequate operability evaluation for the condensate storage tank as required by site procedures. Specifically, upon discovery of the condition, the licensee performed an immediate operability determination evaluation based on concerns with the capability of the loop seal to provide protection from vacuum conditions. Subsequently, the licensee performed additional assessments of their overall program which included the specified operability evaluation in a component design bases review and closure of a confirmatory action letter and failed to identify the inadequacy. During the inspection, the team reviewed the operability determination and identified that the licensee failed to consider or identify concerns with the ability of the condensate storage tank pressure relief valves to operate after a design basis earthquake.

Description. The condensate storage tank is a safety-related tank designed to provide sufficient water for plant shutdown and makeup water for the emergency core cooling system during design basis accidents. The tank is maintained with a slight nitrogen overpressure in order to minimize the water's oxygen concentration and it is provided with two pressure relief valves and a loop seal to protect the tank from overpressure and vacuum conditions. These pressure relief valves are not considered to be safety-related and are classified as seismic category II.

As part of the licensee's component design bases review project conducted in early 2008, the licensee identified concerns with the capability and design basis of the condensate storage tank pressure relief valves and loop seal. In particular, the licensee identified concerns with the capability of the loop seal to provide protection from vacuum conditions as well as the designation of the pressure relief valves as nonsafety-related. The licensee also determined that the design limit of the tank for a vacuum would be exceeded before the loop seal provided any relief.

In response to these concerns, the licensee performed an immediate operability determination to confirm that the pressure relief valves would be capable of protecting the condensate storage tank during design vacuum conditions. The operability determination focused on the capacity of the pressure relief valves during vacuum conditions. The evaluation concluded that the condensate storage tank was operable since the relief valve capacity far exceeded the condensate storage tank vent flow requirements.

During the inspection, the team noted that the operability determination did not address the ability of the pressure relief valves to operate after a design basis earthquake. The operability determination was performed in March 2008 and the licensee had multiple opportunities to identify this oversight. Specifically, the licensee reviewed this operability determination as a part of their confirmatory action letter closure and their component design basis review but still did not identify or address the ability of the pressure relief valves to operate after a design basis earthquake. The team noted that the valves were not considered safety-related and were classified as seismic category II. The team concluded the licensee's operability determination should have addressed the ability of these valves to operate after a seismic event.

In response to the team's concerns, the licensee performed a seismic evaluation for the pressure relief valves. The licensee concluded there was enough margin in the design and construction of the valves to ensure their operation after a design basis earthquake. As a result, the licensee concluded that the condensate storage tank remained operable.

Analysis. The failure to identify an inadequate operability evaluation for the condensate storage tank was a performance deficiency. This finding was more than minor because it was associated with the protection against external events (seismic) attribute of the Mitigating Systems Cornerstone and adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences.

The team evaluated the significance of this finding using Inspection Manual Chapter 0609, Attachment 4, "Phase 1 – Initial Screening and Characterization of Findings." The finding was determined to be of very low safety significance (Green) since the finding did not result in a loss of operability, a loss of system safety function, an actual loss of safety function of a single train for greater than its technical specification allowed outage time, or an actual loss of safety function for greater than 24 hours and the finding did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event.

This finding had a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program since the licensee failed to properly evaluate for operability [P.1.(c)].

Enforcement. Title 10 of the *Code of Federal Regulations*, Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," states, in part:

> Activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings.

Procedure 40DP-9OP26, "Operability Determination and Functional Assessment," requires that the operability determination consider the conditions that could adversely affect the specified safety function of the structures, systems, and components as defined in the current licensing basis, including seismic requirements.

Contrary to the above, on June 11, 2008, the licensee failed to accomplish an activity affecting quality in accordance with their procedures. Specifically, the licensee failed to consider seismic concerns during an operability evaluation of the condensate storage tank pressure relief valves. Because this violation was of very low safety significance and it was entered into the licensee's corrective action program (PVAR 3353683), this violation is being treated as a noncited violation, consistent with Section VI.A.1 of the Enforcement Policy: NCV 05000528; 05000529; 05000530/2009008-02, "Failure to Identify an Inadequate Operability Evaluation for the Condensate Storage Tank."

.3 Results of Reviews for Operating Experience:

.3.1 Inspection of Information Notice 2008-05, Fires Involving Emergency Diesel Generator Exhaust Manifolds, April 12, 2008

a. Inspection Scope

The team reviewed NRC Information Notice 2008-05, "Fires Involving Emergency Diesel Generator Exhaust Manifolds," which documented several instances of small fires on

Fairbanks-Morse diesel generators. Palo Verde has Cooper-Bessemer diesel engines, so the team reviewed the licensee's analysis to ensure that there is no pathway that fuel oil or lube oil could reach the exhaust manifold and render the emergency diesel generator inoperable. The licensee has performed an adequate review of the operating experience with respect to exhaust manifold fires and has concluded that there are two oil lines which have the potential to spray onto the exhaust manifold if they fail. To date, there has not been a failure of either of the two lines, and during daily walkdowns, the licensee ensures that the material condition of the two lines remains in excellent condition.

b. Findings

No findings of significance were identified.

.3.2 NRC Generic Letter 83-28, Required Actions in Response to the Salem Anticipated Transient without a Scram (ATWS) Events

a. Inspection Scope

The team reviewed NRC Generic Letter 83-28, "Required Actions Based on Generic Implications of Salem ATWS Events," which documented required actions in response to the Salem ATWS events. The actions covered by this generic letter fell into four areas: posttrip review, equipment classification and vendor interface, postmaintenance testing, and reactor trip system reliability improvements. The team reviewed the licensee's commitments and their implementation in response to this generic letter. Specifically, the team reviewed the licensee's programs for vendor interface, postmaintenance testing, and reactor trip system reliability improvements associated with the reactor trip breakers.

b. Findings

Failure to Incorporate Vendor Information for Reactor Trip Breakers

Introduction. The team identified a Green noncited violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," with programmatic implications for the licensee's failure to incorporate updated vendor information for the reactor trip breakers. Specifically, the licensee failed to incorporate an updated revision of the maintenance program manual and at least two technical bulletins from the reactor trip breaker vendor.

Description. On February 22 and 25, 1983, anticipated transient without a scram events occurred at the Salem Nuclear Generating Station due to the reactor trip breakers' undervoltage trip attachment sticking as a result of improper maintenance and the failure to incorporate and use vendor information. In response to these events, the NRC issued several generic communications, including Generic Letter 83-28.

These generic communications required that licensees take several actions to ensure the operability of the reactor trip breakers. One required action from Generic Letter 83-28 was for licensees to establish, implement, and maintain a continuing program to ensure that vendor information is complete, current, and controlled throughout the life of the plant, and appropriately referenced or incorporated in plant instructions and procedures.

In a letter dated April 3, 1986, the licensee stated that their procedure "has been changed to ensure that revisions and/or addenda to technical manuals are reviewed by the responsible engineering group for impact on existing, effective station procedures and, where necessary, changes to those procedures implemented as required. This procedure contains all the requirements for controlling vendor technical information throughout the life of the plant, as well as the requirements for permanent plant retention."

The team reviewed the licensee's vendor interface program related to the reactor trip breakers and noted that the licensee used Westinghouse DS-416 breakers for the reactor trip breakers. The team identified that the licensee's vendor technical manual for the Westinghouse DS-416 breaker was last revised in 2003. This vendor technical manual incorporated the guidance contained in Westinghouse maintenance program manual, "MPM-DS Breaker," dated March 31, 1999.

The team also identified that Westinghouse published at least two technical bulletins (TB-04-7, "DS Breaker Fails to Close" and TB-04-8, "DS Cell Switch") and updated the maintenance program manual with Revision 2 since 2003. The inspectors noted that Technical Bulletin TB-04-7 was published on April 6, 2004; Technical Bulletin TB-04-8 was published on May 3, 2004; and the maintenance program manual was revised in May 2008. The team determined that the licensee failed to incorporate this updated information into their vendor technical manual. Specifically, the team noted that the two technical bulletins were not entered into the licensee's site work management system, nor was an engineering document change requested for the maintenance program manual.

In response to the team's concerns, the licensee performed an evaluation of the new vendor information. The licensee determined that the information was applicable to the plant and should be added to the vendor technical manual. The licensee's evaluation also concluded that the new vendor information did not impact the ability of the reactor trip breakers to perform their safety function.

<u>Analysis</u>. The failure to incorporate updated vendor information for the reactor trip breakers was a performance deficiency. This finding was more than minor because it was associated with the equipment performance attribute of the Mitigating Systems Cornerstone and adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. This finding also had the potential, if left uncorrected, to lead to a more significant safety concern since it was similar to the root cause of the Salem anticipated transient without a scram events.

The team evaluated the significance of this finding using Inspection Manual Chapter 0609, Attachment 4, "Phase 1 – Initial Screening and Characterization of Findings." The finding was determined to be of very low safety significance (Green) since the finding did not result in a loss of operability, a loss of system safety function, an actual loss of safety function of a single train for greater than its technical specification allowed outage time, or an actual loss of safety function for greater than 24 hours and the finding did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event.

This finding had a crosscutting aspect in the area of problem identification and resolution associated with operating experience since the licensee failed to implement changes to station processes, procedures, equipment, and training programs [P.2(a)].

Enforcement. Title 10 of the *Code of Federal Regulations*, Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," states, in part:

> Activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings.

Procedure 87DP-0CC08, "Control of Vendor Documentation," describes the licensee's process for receipt and processing of vendor technical information. This procedure requires that all technical bulletins received from the vendor be reviewed to assure that applicability to design documents, configuration control, and maintenance requirements are met. Specifically, this procedure requires that new vendor information affecting safety-related equipment should be processed within 90 days after screening. This procedure also requires that revised or updated vendor information be processed using the engineering document change process. The team noted that the two technical bulletins were not entered into the licensee's site work management system, nor was an engineering document change requested for the maintenance program manual.

Contrary to the above, from April 4, 2004, to July 16, 2009, the licensee failed to accomplish an activity affecting quality in accordance with their procedures. Specifically, the licensee failed to appropriately process vendor information for the reactor trip breakers as required by procedure 87DP-0CC08. Because this violation was of very low safety significance and it was entered into the licensee's corrective action program (PVARs 3354252 and 3355082), this violation is being treated as a noncited violation, consistent with Section VI.A.1 of the Enforcement Policy: NCV 05000528; 05000529; 05000530/2009008-03, "Failure to Incorporate Vendor Information for Reactor Trip Breakers."

.3.3 Generic Letter 95-07 and Information Notice 96-08, Pressure Locking and Thermal Binding of Gate Valves

a. Inspection Scope

The team reviewed Generic Letter 95-07, "Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves," and Information Notice 96-08, "Thermally Induced Pressure Locking of a High Pressure Coolant Injection Gate Valve," which documented several concerns with pressure locking and thermal binding of gate valves. The team reviewed the licensee's response to the generic letter and information notice. Specifically, the team reviewed the licensee's program and calculations to verify the licensee appropriately identified and mitigated safety-related gate valves that were susceptible to pressure locking and thermal binding.

b. Findings

No findings of significance were identified.

.3.4 Information Notice 2004-01, Common Cause Failure of Auxiliary Feedwater Pumps Due to Orifice Fouling

a. Inspection Scope

The team reviewed Information Notice 2004-01, "Auxiliary Feedwater Pump Recirculation Line Orifice Fouling – Potential Common Cause Failure," which documented concerns with the potential common cause failure of auxiliary feedwater pumps due to orifice fouling in recirculation lines. The team reviewed the licensee's response to this information notice. Specifically, the team reviewed the licensee's evaluation under their operating experience program as well as the actions taken to prevent a similar event.

b. Findings

No findings of significance were identified.

.3.5 Inspection of NRC Generic Letter 2007-01, Inaccessible or Underground Power Cable Failures that Disable Accident Mitigation Systems or Cause Plant Transients:

a. Inspection Scope

The team reviewed the generic letter, which documented failures of safety-related cables and their associated systems at several sites due to long-term exposure to moisture. In NRC Generic Letter 2007-01, the NRC requested the status of all cable failures for those cables that were inaccessible or underground as well as a description of inspection, testing, and monitoring programs for these cables. The team reviewed the licensee's response to this generic letter, which reported one cable failure in March 2005 for a safety-related 4160V motor cable for the spray pond pump B for Unit 1. The team also reviewed the related Information Notice 2002-12, which documents several cable failures at several plants due to water intrusion. The team reviewed drawings, cable design and testing specifications, and megger test data. The team reviewed historical data for several electrical vaults and found two vaults with a recent history of water intrusion. The team also reviewed the significant root cause evaluation report for the spray pond cable failure event (CRDR 2784074).

b. Findings

Ineffective Corrective Actions for Vaults Containing Station Blackout Cables

Introduction. The team identified a noncited violation of very low safety significance for failure to effectively implement the corrective action requirements of Regulatory Guide 1.155, "Station Blackout," Appendix A, Criterion 8, "Corrective Action," which were adopted by the licensee in order to meet 10 CFR 50.63, "Loss of All Alternating Current." Although the licensee started a vault monitoring program for water intrusion in vaults with electrical cables in 2003, the effort to prevent exposure of medium voltage cables to submerged conditions has been ineffective for certain vaults that contain the 15Kv station blackout generator output cables. Additionally, there are 27 splices in these cables which have contributed to cable test failures in previous meggar resistance tests that, in some cases, required splice replacement in order to pass the resistance tests.

Description. The NRC issued Regulatory Guide 1.155, "Station Blackout," as an acceptable method for meeting the requirements of 10 CFR 50.63, "Loss of All Alternating Current Power." The Licensee adopted the methodology described in the regulatory guide to comply with 10 CFR 50.63. In response to 10 CFR 50.63, as stated in Arizona Public Service's letter to USNRC, No. 102-05370-CDM/TNW/RAB, dated October 28, 2005, "Revised Station Blackout Evaluation," the licensee adopted Regulatory Guide 1.155, Sections 3.3.5 and 3.5, and Appendix A, as the manner by which they would meet the requirements of 10 CFR 50.63. This includes Appendix A, Criterion 8, "Corrective Action." Criterion 8, "Corrective Action," of Appendix A to Regulatory Guide 1.155 states that "[m]easures should be established to ensure that failures, malfunctions, deviations, defective components and non-conformances are promptly identified, reported, and corrected." The licensee started inspecting and pumping electrical vaults as early as 2003 as a result of Information Notice 2002-12 and the NRC's concern for cable failures due to prolonged submersion in electrical vaults. The cable designs that were used during construction at Palo Verde Nuclear Generating Station did not include the ability to be submerged for the life of the plant. Due to increasing cable failures at commercial nuclear power plants, the NRC issued Generic Letter 2007-01 on submerged cable failures. At that time, the licensee had experienced cable test failures for safety-related and station blackout cables. Of increased concern was the fact that the station blackout cables contain 27 splices, which increase the vulnerability to failure in submerged conditions because these splices are also not designed for continuous submersion for the cable's expected 40 year life. One splice in particular, used because cable length constraints would not allow the normal splice method, utilized a tape and glyptol method of insulating the splice that is even more susceptible to failure than a standard splice.

Electrical vaults AEZV11NKFM48 (vault 48) and AEZV11NKFM49 (vault 49) contain station blackout cables. During a vault inspection on May 30, of 2009, the licensee found 52 inches of water in vault 48. It was pumped out following this inspection. On January 15, 2009, the licensee found 9,180 gallons of water in vault 49. It was also pumped out following this inspection. Vault 49 was inspected again on June 18, 2009, and 35 inches of water were found in the vault. The station blackout generator 15kV cables in these vaults are 18 inches from the bottom of the vault and have experienced repeated exposures to water over the past several years.

Analysis. The failure to effectively prevent exposure to submerged conditions for the station blackout generator 15kV cables was a performance deficiency. The finding is more than minor because it could become a more significant safety concern if left uncorrected and because it is associated with the mitigating systems cornerstone attributes of design control and equipment performance of ensuring the availability, reliability, and capability of safety systems that respond to initiating events.

The team evaluated the significance using Inspection Manual Chapter 0609, Attachment 4, "Phase 1 – Initial Screening and Characterization of Findings." The finding was determined to be of very low safety significance (Green) since the finding did not result in a loss of operability, a loss of system safety function, an actual loss of safety function of a single train for greater than its technical specification allowed outage time, or an actual loss of safety function for greater than 24 hours and the finding did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event.

The finding was reviewed for crosscutting aspects and none were identified.

Enforcement. Title 10 of the *Code of Federal Regulations*, Part 50, Section 63, "Loss of All Alternating Power," requirements are met by the licensee through commitment to Criterion 8, "Corrective Action," of Appendix A to Regulatory Guide 1.155 which states, in part:

> [M]easures should be established to ensure that failures, malfunctions, deviations, defective components and non-conformances are promptly identified, reported, and corrected.

Contrary to the above, as of June 18, 2009, appropriate measures were not established to ensure that failures, malfunctions, deviations, defective components and non-conformances were promptly identified and corrected. Specifically, the station blackout generator 15kV output cables for both generators were exposed to submerged conditions for which they were not designed for extended periods of time. Because the finding is of very low safety significance (Green) and has been entered into the licensee's corrective action program (PVAR 3350712, 3350713, 3350939, and 3352858, this violation is being treated as a noncited violation, consistent with Section VI.A.1 of the Enforcement Policy: NCV 05000528, 05000529, 05000530/2009008-04, "Inadequate Corrective Actions for Vaults Containing SBO Generator Cables."

.4 **Results of Reviews for Operator Actions:**

The team selected risk-significant components and operator actions for review using information contained in the licensee's probabilistic risk assessment. This included components and operator actions that had a risk achievement worth factor greater than or equal to 1.3 or a Birnbaum value greater than 1E-6.

a. Inspection Scope

For the review of operator actions, the team observed operators during simulator scenarios associated with the selected components as well as observing simulated actions in the plant using job performance measure techniques.

Inspection Procedure 71111.21 requires a review of three to five relatively high-risk operator actions. The sample selection for this inspection was five operator actions.

The selected operator actions were:

- Auxiliary operators, working in parallel during real time for a station blackout event, must 1) manually align backup nitrogen supplies to the atmospheric dump valves, 2) start and align the station blackout generators for electrical supply to the safety-related buses (also referred to as the gas turbine generators), and 3) manually realign breakers and trips in the safety-related switchgear (all job performance measures were performed as one action during the previous component design basis inspection and were not achievable during the 60 minutes assumed in the risk analysis)

- Auxiliary operator must manually align the turbine-driven auxiliary feed water pump AFA-P01 during a loss of all power to the M41 bus (job performance measure)

- Control room staff must align the nonsafety auxiliary feedwater pump AFN-P01 and restore feed to the steam generators given a loss of offsite power with complete loss of normal feed water and a failure of both safety-related auxiliary feedwater pumps (Scenario)

- Control room staff must recover the remaining emergency diesel generator given a loss of offsite power with 1 emergency diesel generator out of service and the last emergency diesel generator does not auto-close on the emergency bus with the potential for station blackout (Scenario)

- Control room staff must emergency borate during an anticipated transient without scram event during the first 15 minutes of the event (Scenario)

b. Findings

No findings of significance were identified.

4 OTHER ACTIVITIES

4OA2 Identification and Resolution of Problems

a. Inspection Scope

The team reviewed a sample of problems that the licensee had identified previously and entered into the corrective action program. The team reviewed these issues to verify an appropriate threshold for identifying issues and to evaluate the effectiveness of corrective actions. In addition, condition reports written on issues identified during the inspection were reviewed to verify adequate problem identification and incorporation of the problem into the corrective action system. The specific documents that were sampled and reviewed by the team are listed in the attachment.

b. Findings

No findings of significance were identified.

4OA6 Meetings, Including Exit

On July 16, 2009, the team leader presented the preliminary inspection results to Mr. Hesser, Vice President - Engineering, Palo Verde Nuclear Generating Station, and other members of the licensee's staff.

On October 8, 2009, the inspection team leader conducted a telephonic final exit meeting with Mr. Bement, Site Vice President, and other members of the licensee's staff. The licensee acknowledged the findings during each meeting. While some proprietary information was reviewed during this inspection, all was returned and no proprietary information was included in this report.

4OA7 Licensee Identified Violations

None.

Attachments: Supplemental Information

ATTACHMENT

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee personnel

S. Bower, Nuclear Regulatory Affairs, Director
J. Hesser, Engineering, Vice President
K. Chavet, Nuclear Regulatory Affairs, Senior Consultant
M. Coevin, Operations, Department Leader
E. Dutton, Nuclear Assurance Department, Director
T. Engbring, Design Engineering, Senior Engineer
D. Elkington, Nuclear Regulatory Affairs, Senior Consultant
E. Gouvier, Electrical Design, Senior Engineer
T. Hook, Probabilistic Risk Assessment, Section Leader
M. Karbassian, Engineering/Component Design Bases Review, Director
C. Karlsson, Design Engineering, Section Leader
M. Lacal, Performance Improvement Department, Director
H. Leake, Electrical Design, Senior Consultant Engineer
P. Paramithas, Design Engineering, Acting Director
M. Prieve, Nuclear Security, Emergency Services Director
J. Molden, Engineering Training, Director
F. Oreshack, Nuclear Regulatory Affairs, Compliance Consultant
J. Proctor, Nuclear Regulatory Affairs, Section Leader
J. Rowland, Nuclear Fuels, Section Leader
S. Smith, Operations, Senior Technical Advisor
J. Summy, Engineering, Director
B. Thiele, Component Design Bases Review, Department Leader
M. Van Dop, Design Engineering, Department Leader
J. Weber, Regulatory Affairs, Department Leader
J. Wood, Training, Department Leader

LIST OF ITEMS OPENED AND CLOSED

OPEN

05000528; -529; -530/2009008-01	URI	Calculations to support the graded Voltage Relay Setpoint (Section .2.14)

Opened and Closed

05000528; -529, -530/2009008-02	NCV	Failure to Perform an Adequate Operability Evaluation for the Condensate Storage Tank (Section .2.16)
05000528; -529, -530/2009008-03	NCV	Failure to Incorporate Vendor Information for Reactor Trip Breakers (Section .3.2)
05000528; -529; -530/2009008-04	NCV	Inadequate Corrective Actions for Vaults Containing SBO Generator Cables (Section 3.5)

Corrective Action Documents

CRAI 3194025	CRAI 3251876	CRAI 3264525
CRDR 2784074	CRDR 2780693	CRDR 77868
CRDR 3007513	CRDR 3130922	CRDR 2840920
CRDR 3051228	CRDR 3311031	CRDR 2910912
CRDR 3121410	CRDR 3264869	CRDR 2914478
CRDR 3130903	CRDR 2803441	CRDR 79337
CRDR 3165478	CRDR 3187821	CRDR 2838368
CRDR 3211406	CRDR 2821210	CRDR 950607
CRDR 3249924	CRDR 2839237	CRDR 950836
CRDR 3249927	CRDR 2970059	CRDR 2803441
CRDR 3252469	CRDR 2667701	CRDR 3339346
CRDR 3256729	CRDR 3193970	CRDR 2835132
CRDR 3282707	CRDR 3192744	CRDR 2838079
CRDR 3285804	CRDR 2782680	CRDR 3340180
CRDR 9-9-0847		
PVAR 3191963	PVAR 3188067	PVAR 3353116
PVAR 3267413	PVAR 3266146	PVAR 3350713
PVAR 3091001	PVAR 3295356	PVAR 2843631
PVAR 3203617	PVAR 3256160	PVAR 3350712
PVAR 3253612	PVAR 3223247	PVAR 2980758
PVAR 3251660	PVAR 3222864	PVAR 3352858
PVAR 2949167	PVAR 3256160	PVAR 2982699
PVAR 2960946	PVAR 3225531	PVAR 2985197

PVAR 3019401	PVAR 3005342	PVAR 3228646
PVAR 3036970	PVAR 3039250	PVAR 3350939
PVAR 3214280	PVAR 3041864	PVAR 3351510
PVAR 3230961	PVAR 3044746	PVAR 3245075
PVAR 3232689	PVAR 3072557	PVAR 3274141
PVAR 3240386	PVAR 3128794	PVAR 3342178
PVAR 3242362	PVAR 3131569	PVAR 3346689
PVAR 3244047	PVAR 3141155	PVAR 3280971
PVAR 3252655	PVAR 3155366	PVAR 3350232
PVAR 3345438	PVAR 3168297	PVAR 3302355
PVAR 3212013	PVAR 3187026	PVAR 3187290
PVAR 2978726	PVAR 3187290	PVAR 3188566

PVAR 3353611*	PVAR 3346795*	PVAR 3354056*
PVAR 3354009*	PVAR 3350317*	PVAR 3354252*
PVAR 3352797*	PVAR 3351422*	PVAR 3355082*
PVAR 3353683*	PVAR 3351791*	PVAR 3352271*
PVAR 3352805*	PVAR 3354296*	PVAR 3353007*
PVAR 3351602*	PVAR 3346822*	PVAR 3353696*
PVAR 3347440*		

* Indicates PVARs generated as a result of the CDBI

Calculations

01-EC-MA-0221, AC Distribution, Revision 11

01-EC-PH-0253, 120VAC Distribution, Revision 9

01-EC-PH-0255, 120VAC Control Circuits, Revision 1

13-EC-PB-0101, Undervoltage Protection, Revision 9

13-EC-PB-0202, 4160 V Degraded Voltage Relay Setpoint & Calibration, Revision 03

13-EC-PB-0203, Motor Starting Times, Revision 4

01-EC-MA-0221, AC Distribution Engineering Calculation, Revision 11

02-EC-MA-0221, AC Distribution Engineering Calculation, Revision 13

03-EC-MA-0221, AC Distribution Engineering Calculation, Revision 10

13-JC-AF-0205, Turbine Driven AF Pump Control Settings, Revision 5

13-JC-CT-0200, Setpoints and Total Loop Uncertainty for High/Low Condensate Tank Levels (Loops CTALLOOP0035 and CTBLLOOP0036), Revision 9

13-MC-AF-0800, Auxiliary Feedwater ESF Function Response Times, Revision 7

13-MC-CT-0205, Condensate Storage Tank, Revision 6

13-MC-CT-0302, CST Relief Valves, Revision 1

13-MC-CT-0307, CST Minimum Level Setpoint, Revision 5

01-JC-ZZ-0223, 89-10 Program Motor Operated Valve Adjusted Setpoint Calculation, Revision 3

02-JC-ZZ-0223, 89-10 Program Motor Operated Valve Adjusted Setpoint Calculation, Revision 4

03-JC-ZZ-0223, 89-10 Program Motor Operated Valve Adjusted Setpoint Calculation, Revision 3

13-JC-ZZ-201, MOV Thrust, Torque and Actuator Sizing Calculation, Revision 14

13-MC-GA-0211, Backup Nitrogen Cylinder Determination for Additional Station Blackout Coping Time and Relief Valve Size Determination for High Pressure Nitrogen Piping, Revision 1

13-MC-HA-0052, Auxiliary Building Essential Cooling System Heat Load Calculation, Revision 7

13-MC-SI-0017, Safety Injection System Interface Requirements Calculation, Revision 6
13-MC-SI-0215, HPSI System Performance Evaluation and Surveillance Requirement Basis Calculation, Revision 4

13-MC-SI-0240, Low Pressure Safety Injection System Hydraulic Performance Analysis and System Surveillance Criteria, Revision 0

13-MC-SI-318, LPSI/CS Miniflow Check Valve Failure Analysis, Revision 0

13-MC-SG-0211, AOV Thrust and Actuator Sizing Calculation - CCI Drag Valves, Revision 2

13-MC-SG-405, ADV Nitrogen Tank Temperature Adjusted Pressures, Revision 2

Design Basis Documents

AF, Auxiliary Feedwater System, Revision 18
CD, Condensate System, Revision 8
SG, Main Steam, Revision 23
Feedwater System DBM, Revision 11
Main Steam DBM, Revision 23
Safety Injection System, Revision 29

Drawings

01-E-PBA-001, Single Line Diagram 4.16 kV Class 1E Power System Switchgear 1E-PBA-S03, Revision 8

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